NON-PUBLIC?: N
ACCESSION #: 9103130246
LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK - UNIT 1 PAGE: 1 OF 07

DOCKET NUMBER: 05000445

TITLE: REACTOR TRIP CAUSED BY PERSONNEL ERROR AND INSUFFICIENT LABELING
OF SENSITIVE EQUIPMENT
EVENT DATE: 02/10/91 LER #: 90-004-00 REPORT DATE: 03/12/91

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 080

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: T.A. HOPE COMPLIANCE SUPERVISOR TELEPHONE: (817) 897-6370

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On February 10, 1991, Comanche Peak Steam Electric Station Unit 1 was in Mode 1, Power Operation, at 80 percent reactor power. While conducting on-the-job training, an Auxiliary Operator opened the 6.9 KV switchgear bus "1A3" auxiliary cabinet potential transformer drawer resulting in a load shed signal for the bus loads, which included Reactor Coolant Pump 1-03, followed by a reactor trip due to a loss of flow in reactor coolant loop 3. The root causes were determined to be inadequate labeling on the potential transformer drawer and poor judgement in opening a drawer without knowing the effect it would have on the plant. Corrective actions include labeling the potential transformer drawers, identifying equipment or panels that have interlocks or special sensitivity for equipment actuation, and issuing guidance to personnel on opening switchgear doors.

END OF ABSTRACT

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System.

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On February 10, 1991, just prior to the event, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, with reactor power at 80 percent.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems or components that contributed to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

At 0530 on February 10, 1991, an Auxiliary Operator (AO) (utility, non-licensed) was conducting on-the-job training with a trainee (utility, non-licensed). The AO and trainee were trying to locate the reset switches for the Circulating Water Pumps 1-03 and 1-04 (EIIS:(KE)(P)). While looking for the reset switch for Circulating Water Pump 1-03, the AO opened the potential transformer drawer (EIIS:(EA)(FD)) at the bottom of the switchgear bus "1A3" auxiliary cubicle (EIIS:(EA)(BU)). The AO did not realize that the drawer was position interlocked with fuses for the bus undervoltage load shed circuitry (EIIS:(EA)(RLY 94)). There was no warning or caution label on the drawer. Opening the drawer resulted in a load shed signal for the bus loads, which included Reactor Coolant Pump 1-03 (EIIS:(AB)(P)), followed by a reactor trip due to loss of flow in reactor coolant loop 3. Also lost when the drawer was opened were Condensate Pump 1-01 (EIIS:(SG)(P)), Turbine Plant Cooling Water Pump 1-01 (EIIS:(KB)(P)) and Service Air

Compressor 1-01 (EIIS:(LF)(CMP)). Circulating Water Pump 1-03 (EIIS:(KE)(P)) would also have tripped if it had been in service. Following the trip, Control Room personnel responded in accordance with

emergency operating procedures. Plant systems responded as expected, except for the situations discussed below. The plant was stabilized in Mode 3, Hot Standby, at 0632. At 0720 the NRC was notified of the event via the Emergency Notification System in accordance with 10CFR50.72.

Following the reactor trip, three momentary Steam Generator level signals (EIIS:(JE)(RLY 83)) were received. LO-LO Level signals were received for Steam Generators 2 and 3 at approximately 1.69 seconds and a HI-HI Level signal was received for Steam Generator 2 at approximately 2.18 seconds after the trip. The Turbine Driven Auxiliary Feedwater Pump (EIIS:(BA)(P)) did not start and Steam Generator Blowdown and Sampling (EIIS:(KN)(RLY 56)) did not isolate. A Main Feedwater isolation signal was generated as a result of the HI-HI Level signal with Main Feedwater Pump 1A tripping (EIIS:(SJ)(P)). Main Feedwater Pump 1B did not trip. The steam dump turbine load rejection interlock (EIIS:(JI)(IEL)) would not reset.

The momentary Steam Generator level signals had been previously identified in another event as the result of steam pressure oscillations following a turbine trip. Two channels on all four Steam Generators are susceptible to low level spikes and two channels on Steam Generators 1 and 2 are susceptible to high level spikes. The level transmitters share the same impulse lines as the steam flow transmitters and as a result, the narrow range level transmitters are susceptible to level spikes whenever rapid changes in steam flow occur. Due to the short duration of the spikes, less than 0.2 seconds, the signals were not present long enough to start Turbine Driven Auxiliary Feedwater Pump, isolate Steam Generator Blowdown and Sampling, or trip Main Feedwater Pump 1B. During the post trip recovery the slave relays for these signals were tested and showed that both trains were satisfactory and would respond as designed to actual Steam Generator LO-LO or HI-HI signals. A bad Lead/Lag card was found in the Steam Dump Turbine Load Rejection circuitry which was replaced and calibrated with all alarms functioning and the interlock resetting correctly.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE OR
PROCEDURAL ERROR

The reactor trip was annunciated by numerous alarms in the Control Room. The immediate cause of the trip was reported by the AO.

II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

No failed components contributed to this event.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

No failed components contributed to this event.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF
COMPONENTS WITH MULTIPLE FUNCTIONS

No failed components contributed to this event.

D. FAILED COMPONENT INFORMATION

No failed components contributed to this event.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The Reactor Protection System (EIIS:(JC)) and Auxiliary Feedwater System (EIIS:(BA)) actuated during the event; all associated components within these systems functioned as designed. The Steam Generator level signals received immediately after the trip were not valid signals, as explained above, and did not affect the safe shutdown or recovery of the plant.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

No safety system trains were inoperable as a result of this event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The reactor trip was the result of the loss of Reactor Coolant Pump 1-03. The reactor trip event is discussed in Section 15.3.1 of the CPSES Final Safety Analysis Report under "Partial Loss of Forced Reactor Coolant Flow." The analysis uses conservative assumptions to demonstrate that Departure from Nucleate Boiling Ratio (DNBR) will never decrease below the limiting value of 1.30 during the event. The event of February 10, 1991, occurred at 80 percent reactor power, and all protective functions responded as designed. The event is completely bounded by the FSAR accident analysis which assumes an initial power level of 102 percent and makes conservative assumptions which reduce the capability of safety systems to mitigate the consequences of the transient. The event of February 10 did not adversely affect the safe operation of CPSES Unit 1 or the health and safety of the public.

IV. CAUSE OF THE EVENT

A. ROOT CAUSE

1. The 6.9 KV switchgear bus "1A3" auxiliary cabinet potential transformer drawer did not have an adequate label. This drawer contains a vital position interlock with bus undervoltage load shedding circuitry, but was not labeled as such.

2. The AO used poor judgement in opening a drawer without knowing the effect it would have on the plant. Operators are expected to know the result of any action they take in the plant.

B. GENERIC CONSIDERATIONS

Similar equipment needs to be identified and labeled to prevent similar events from occurring.

V. CORRECTIVE ACTIONS

CORRECTIVE ACTIONS TO PREVENT RECURRENCE

A. ROOT CAUSE

1. The 6.9 KV switchgear bus "1A3" auxiliary cabinet potential transformer drawer did not have an adequate label.

CORRECTIVE ACTION

Labels have been affixed to the 6.9 KV switchgear bus "1A1", "1A2", "1A3", and "1A4" auxiliary cabinet potential transformer drawers.

2. The AO used poor judgement in opening a drawer without knowing the effect it would have on the plant. Operators are expected to know the result of any action they take in the plant.

CORRECTIVE ACTION

Operations Management has issued a memorandum that provides additional guidance on opening switchgear doors. This memorandum has been included in the Control Room "Lessons Learned" notebook.

B. GENERIC CONSIDERATIONS

Similar equipment needs to be identified and labeled to prevent similar events from occurring.

CORRECTIVE ACTION

A survey will be conducted to identify equipment that could cause similar problems. The equipment will be labeled appropriately.

VI. PREVIOUS SIMILAR EVENTS

There have been no previous reactor trips attributed to the causes identified during this event investigation. No previous reactor trips have been caused by personnel taking action in the plant that was not covered by a procedure or work control process, nor have any previous reactor trips been caused by insufficient labeling of plant equipment.

VII. ADDITIONAL INFORMATION

The times listed in the report are approximate and Central Standard
Time.

ATTACHMENT 1 TO 9103130246 PAGE 1 OF 1

Log # TXX-91077
File # 10200
910.4
Ref. # 50.73(a)(2)(iv)
TUELECTRIC
March 12, 1991

W. J. Cahill
Executive Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
MANUAL OR AUTOMATIC REACTOR PROTECTION SYSTEM ACTUATION
LICENSEE EVENT REPORT 91-004-00

Gentlemen:

Enclosed is Licensee
Event Report 91-004-00 for Comanche Peak Steam
Electric Station Unit 1, "Reactor Trip Caused by Personnel Error and
Insufficient Labeling of Sensitive Equipment."

Sincerely,

William J. Cahill, Jr.
JAA/daj

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

400 North Olive Street LB 81 Dallas, Texas 75201

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